FENOC
FirstEnergy Nuclear Operating Company

76 South Main Street
Akron, Ohio 44308

Lew W. Myers
Chief Operating Officer

330-384-3733
Fax: 330-384-3799

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United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
LER 2004-001-01

Ladies and Gentlemen:

Enclosed is Licensee Event Report (LER) Supplement 2004-001-01, Emergency Service Water Pump Failure. This supplement is being submitted to update the root cause, to update the additional corrective actions and to provide 10CFR21 required information.

There are no regulatory commitments contained in this letter. Any actions discussed in this document that represent intended or planned actions, are described for the NRC's information, and are not regulatory commitments.

If you have questions or require additional information, please contact Mr. Jeffrey J. Lausberg, Manager – Regulatory Compliance, at (440) 280-5940.

Very truly yours

Lew W. Myers

Enclosures: LER 2004-001-01

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 6/30/2007

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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4. TITLE
Emergency Service Water Pump Failure

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
05	22	2004	2004 -	001 -	01	09	18	2004	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 1

10. POWER LEVEL: 100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☒ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☒ 50.73(a)(2)(ii)(a)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☒ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☒ 50.73(a)(2)(i)(A)	☒ 50.73(a)(2)(v)(C)	☒ OTHER Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☒ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER *(Include Area Code)*
Kenneth Russell, Compliance Engineer, Regulatory Compliance	(440) 280- 5580

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	BI	P	G200	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete EXPECTED SUBMISSION DATE).*	☒ NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On May 21, 2004, at 0150 hours with the Perry Nuclear Power Plant operating in MODE 1 at 100 percent power, Emergency Service Water (ESW) pump A was declared inoperable due to pump shaft coupling failure. At 1600 on May 21, 2004, a power reduction was ordered to shutdown the plant to complete repairs. MODE 3 (hot shutdown) was entered at 2139 on May 22, 2004. MODE 4 (cold shutdown) was entered at 0619 on May 23, 2004. Although bypassing of steam to the main condenser was available, an alternate decay heat removal system was not established in MODE 4 within the Technical Specification (TS) 3.4.10 completion time.

The failure mechanism of the coupling was stress corrosion cracking (SCC). This was the second occurrence of this type of failure on the same pump in less than one year. Since the ESW pump B could not be physically inspected, an engineering evaluation for the susceptibility of ESW pump B to SCC could not support continued operability. At 1500 on May 24, 2004 ESW pump B was declared inoperable.

The stress corrosion cracking failure mode was due to marginal coupling design, insufficient coupling keyway corner radius, and susceptible environmental conditions. The ESW pump B continued to operate until removed from service for inspection and upgrade to a new coupling design. Both A and B ESW pump couplings were replaced with couplings constructed of less susceptible material and additional design margin. This event is considered to be of very low safety significance.

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INTRODUCTION

The emergency service water (ESW) [BI] system at the Perry Nuclear Power Plant (PNPP) is comprised three independent subsystem loops A, B and C. These open subsystem loops take suction from and return to Lake Erie. Each loop is supplied by a separate pump, which is operated from a preferred power source or a standby diesel generator (DG) [EK]. The ESW loops A & B supply cooling water to residual heat removal (RHR) [BO], standby DG and emergency closed cooling (ECC) [CC] water heat exchangers [HX], plus a fuel pool heat exchanger [DA-HX]. The C loop supplies cooling water to the high pressure core spray (HPCS) [BG] diesel generator and the HPCS pump room cooler [CLR]. The ESW loops run intermittently for plant evolutions such as waste discharges, chemical treatments, ESW testing and testing of supported systems. The ESW loops only normally run for extended periods to support plant operations such as shutdown cooling during outages and when necessary to support operation of control complex chillers and ventilation trains.

On May 21, 2004, at 0150 hours with the PNPP operating in MODE 1 at 100 percent power, ESW pump A [BI-P] was declared inoperable due to indications that the pump had failed. It was later determined that a pump shaft coupling had failed. As a result of ESW A inoperability, Limiting Condition of Operation for Technical Specifications (TS) 3.7.1, Emergency Service Water (ESW) System - Divisions 1 and 2, and 3.8.1, AC Sources-Operating, were entered. This required the pump to be returned to operable within 72 hours or be in hot shutdown in the following 12 hours and cold shutdown in 36 hours. The decision was made to place the plant in cold shutdown to complete the repairs.

EVENT DESCRIPTION

The ESW A pump was started at 0148 hours on May 21, 2004. At 0150 hours, after two (2) minutes operation, ESW A pump (Gould model VIT 20X30 BLC, 2 stage, 800 hp, 1185 rpm) indicated a loss of flow. The control room staff observed ESW A flow indications at 0 gallons per minute (gpm) with the motor still running, which indicated a pump shaft failure. A trouble shooting and decision making team was established to provide input to the operating crew.

At 1602 hours on May 21, 2004, a power reduction was commenced to shutdown the plant to MODE 4. Commencing a TS required shutdown requires that the Nuclear Regulatory Commission (NRC) be notified accordance with 10CFR50.72(b)(2)(i). The required notification was completed via ENF 40767 within required 4 hour time allowance, at 1611 hours.

The power decrease continued until MODE 3 (hot shutdown) was entered at 2139 hours on May 22, 2004. Completion of a TS required shutdown, entry into MODE 3, requires the submittal of a Licensee Event Report (LER) per 10CFR50.73(a)(2)(i)(A).

At 0014 hours on May 23, 2004, the reactor pressure decreased below the RHR cut in permissive pressure (135 psig) making the requirements of TS 3.4.9 applicable. The Required Actions for this LCO are to initiate action to restore RHR shutdown cooling subsystems to operable immediately or verify an alternate method of decay heat removal (ADHR) is available for each inoperable RHR shutdown cooling subsystem and be MODE 4 (cold shutdown) in 24 hours. The ADHR method was determined to be bypass of steam to the condenser.

TS 3.7.1 required the plant to be placed in MODE 4, which was achieved at 0619 hours on May 23, 2004. Once in MODE 4, TS 3.4.10, Residual Heat Removal (RHR) Shutdown Cooling System-Cold Shutdown, requires that an ADHR system be available for each inoperable shutdown cooling loop of RHR. Use of the main condenser for steam cooling is not available in MODE 4. Heating up above 200 degrees for steam

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cooling to the condenser would result in a mode change that would not be permitted by TS due to the inoperable equipment. Since an ADHR method that would maintain MODE 4 was not demonstrated for RHR A within the required action completion time, a condition prohibited by TS occurred and is reportable per 10CFR50.73(a)(2)(i)(B). ESW pump A was returned to operable status at 0513 hours on May 29, 2004.

During the investigation of the ESW pump A coupling failure, a concern was raised that the failure mechanism could also exist for the ESW pump B. Both pumps had identical coupling design and similar run times. An operability determination (OD) was requested to address the potential for a common cause failure of the ESW pump B. The control room staff was notified that the response to the requested OD could not support continued operability of the ESW pump B. As a result the Shift Manager declared the ESW pump B inoperable at 1500 hours on May 24, 2004. However, this pump continued to operate, removing decay heat, until removed from service. Since ESW pump A was previously inoperable, this resulted in both ESW pumps A and B being inoperable which was determined to be a loss of the safety function. This condition was reported as an 8 hour notification per 10CFR50.72(b)(3)(v)(B) and is an LER per 10CFR50.73(a)(2)(v)(B),(C),(D).

Concurrent inoperability of the ESW pumps A and B was also determined to be reportable as a seriously degraded condition since suppression pool cooling was not available to maintain the containment barrier integrity. This condition is reportable per 10CFR50.72(B)(3)(ii)(A) and as an LER per 10CFR50.73(a)(2)(ii)(A). The required notifications were made via ENF 40774 within the required 8 hours at 1712 hours on May 24, 2004.

Since ESW pump A coupling failed and it could not be assured that the ESW pump B coupling was not susceptible to the same failure mechanism the condition was determined to be reportable per 10CFR50.73(a)(2)(vii), Common-Cause Inoperability of Independent Trains. There is no corresponding 10CFR50.72 reporting requirement.

Although the ESW pump B was considered to be inoperable, it remained in operation for about 3 days while the ESW pump A was repaired with upgraded couplings. ESW pump A was returned to operable at 0513 hours on May 29, 2004.

Following the ESW pump A restoration, the ESW pump B was modified with upgraded couplings. The ESW pump B was declared operable following coupling replacement at 1523 on June 3, 2004.

CAUSE OF EVENT

The analysis of the ESW pump A couplings determined that the couplings had failed from stress corrosion cracking (SCC). There are three parameters that must be present for the ESW couplings to fail by the mechanism of SCC; susceptibility of material for SCC, a corrosive environment, and a stress intensity that exceeds the threshold for SCC on the pump shaft coupling. Inherent to the Perry installation is the susceptible material (416 SS) and the corrosive environment (lake water).

The following are the technical (hardware) causes:

- Less Than Adequate Coupling Design

The original coupling design has a coupling sleeve outer diameter of 4.0 inches and is manufactured out of 416 stainless steel (ASTM A582), heat treated to 1000°F. This size coupling provides a design wall thickness, based on machining tolerances, ranging between 0.336 and 0.347 inches. The difference between actual and allowable shear stresses (based ASME B106.1M-85 "Design of Transmission Shafting") was marginal. This condition has been determined to be reportable under 10CFR Part 21. The information specifically required to be submitted by this regulation is included later in this report.

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- Insufficient Coupling Keyway Corner Radius

Design coupling sleeve keyway corner radius is 0.030 to 0.040 inches. The failed coupling showed an existing radius of the #1 coupling keyway corner of 0.01 inches. The reduction in the keyway radius has a significant impact on the stress concentration which translates to the applied stress. This could not be proven to be manufacturer or assembly techniques.

- Forebay Environment

An environment conducive for stress corrosion cracking is necessary for the failure mechanism. The ESW pumps take suction from the forebay that is fed from Lake Erie. Lake Erie water chemistry includes 20 - 30 parts per million (ppm) of chlorides. The pumps are subjected to monthly chemical treatment for biological control, intermittently increasing the chloride level to 30 - 40 ppm. The location of the first two couplings closest to the pump-to-motor rigid adjustable coupling are above the normal sump level. As such, these two couplings are out of the environment when the pumps are not running. This has the potential of concentrating contaminants in the coupling/shaft interface region making it potentially more susceptible to SCC.

Organizational and management issues were identified during the root cause evaluation. These issues contributed to not identifying the significance of the coupling design weakness earlier. Additionally, quality issues resulting in dimensional deficiencies were identified for parts supplied by the coupling vendor.

The ESW pump B had been operating as needed since it's last rebuild in April of 2003. Prior to that rebuild the pump had operated effectively for approximately five (5) years. Additionally, the ESW pump B coupling was sent for analysis following removal. No cracks were found; however, there was insufficient evidence to conclude that the ESW B couplings were not subject to the same failure mechanisms as the ESW pump A couplings. The presence of random incipient corrosion in the removed ESW pump B coupling #1 and #2 indicated that the conditions were conducive for the onset of stress corrosion cracking. With the data available, it is difficult to predict how long it would have taken for stress corrosion cracks to initiate and propagate to the point that coupling failure would occur.

EVENT ANALYSIS

The following systems were considered inoperable as a result of the failure of ESW A pump and the inoperability of ESW pump B.

- Low Pressure Core Spray
- Division 1 and 2 Emergency Diesel Generators
- Reactor Core Isolation Cooling
- Residual Heat Removal A, B, and C
- Control Room HVAC Train A and B
- Emergency Pump Area Cooling A and B
- Control Complex Chill Water A and B
- Hydrogen Analyzer A and B
- Emergency Closed Cooling A and B
- Combustible Gas Mixing Compressor A and B

The ESW C subsystem was operable and available to perform its design function to cool division 3 loads.

The safety analysis was divided into three parts consisting of ESW pump A failure at power, ESW pump A failure while shutdown and failure of ESW pump A with potential for failure of ESW pump B.

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ESW pump A was determined to be inoperable and unavailable to perform its intended function on May 21, 2004. The plant was taken to MODE 2 at 1449 hours on May 22, 2004. Shutdown cooling was initiated in MODE 3 on May 23, 2004 at 0307 hours.

The mission time for successful operation of the ESW pumps in the PSA model is 24 hours. ESW pump A ran for over 24 hours during maintenance on the Control Complex Chillers (P47) from April 24, 2004 until 1732 hours on May 13, 2004. Between May 13 and 21 ESW A ran for an additional 11 hours. It was assumed that ESW A would have been able to perform its intended function if called upon prior to May 13, 2004. However, from May 13, 2004 until the plant was shutdown, it is questionable whether ESW A would have been able to operate for 24 hours. From about 0435 hours on May 13 until shutdown cooling was initiated on May 23 at 0307 hours it is assumed that ESW pump A would not have been able to run for its 24-hour mission time.

The Incremental Conditional Core Damage Probability (ICCDP) associated with the unavailability of ESW A for 10 days (May 13 through May 23) was computed using the PSACY10 PRA model.

Summary of Results:

1. The failure of ESW pump A for 10 days results in an ICCDP of 6.8E-07.

2. The seismic ICCDP due to ESW A being unavailable for 10 days is about 3.3E-08. The ICCDP due to fires is about 5.8E-08. The contribution due to external flooding is considered to be non-significant. The ICCDP including all external events is approximated to be 7.7E-07.

3. The failure of ESW pump A for 10 days also has an impact on the large early release probability. The Incremental Conditional Large Early Release Probability (ICLERP) due to the unavailability of ESW pump A for 10 days is less than 1.0E-07.

The evaluation with ESW A unavailable for 10 days at power indicates the event significance was of very low safety significance.

A separate calculation was performed that covers the period when the pressure in the reactor vessel was less than 135 psig and shutdown cooling (by RHR) was established until both RHR subsystems were available for shutdown cooling and suppression pool cooling, and the Division 1 Diesel Generator became available.

A total of 5 accident sequences (Loss Of RHR - 3, Loss Of Offsite Power (LOOP) - 2) were reviewed, taking into consideration the unavailability of ESW pump A and its supported systems given a loss of the operating RHR train or a LOOP. Using the prescribed methodology from the Counting Rule Worksheet the risk significance characterization for this event during the exposure time window for shutdown operations is of very low safety significance.

A third condition, ESW pump A failure with potential for failure of ESW pump B, was evaluated in accordance with NUREG/CR-6819, Vol. 3 (INEEL/EXT-99-00613), Common Cause Failure- Event Insights. The NUREG lists four criteria that must be met for an event to be classified as resulting from a common-cause. Review of the criteria determined that this event did not require evaluation as a common-cause failure.

This event has been evaluated to be of very low safety significance.

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10CFR PART 21 REQUIRED INFORMATION

1) Name and address of the individual or individuals informing the commission.

Lew W. Myers, Chief Operating Officer
Perry Nuclear Power Plant
10 Center Road
P.O. Box 97
Perry, Ohio 44081

2) Identification of the facility, the activity, or the basic component supplied for such facility or such activity within the United States which fails to comply or contains a defect.

Emergency Service Water (ESW) Pump shaft coupling - The couplings are Gould pump part number C3261-6-2265 and are made to A-582-95b UNS S41600 of A582-95b, Type 416 (UNS S41600) Martensitic Stainless Steel. The mill certification for heat #72479 stated that the coupling material was supplied by Atlas Specialty Steel Inc.

3) Identification of the firm constructing the facility or supplying the basic component which fails to comply or contains a defect.

The coupling that failed in September 2003 was supplied by Goulds under their 10 CFR 50 Appendix B Program. The coupling that failed in May 2004 is a commercial item manufactured by Goulds, dedicated and supplied by Enertech under their 10 CFR 50 Appendix B Program.

4) Nature of the defect or failure to comply and the safety hazard which is created or could be created by such defect or failure to comply.

Description of Deviation or Failure to Comply
The nature of the deviation is that a critical component, the ESW pump shaft sleeve coupling, is marginally designed. The coupling is constructed of A-582 Type 416 stainless steel material. The material from which the coupling is constructed is tempered to obtain elevated hardness, however, the higher hardness increases the materials susceptibility to intergrannular stress corrosion cracking (IGSCC). The operating stress in the installed coupling is significantly high such that high transient motor start-up torque and improper tolerance of parts result in stress levels sufficient to promote stress corrosion cracking of the coupling.

The critical characteristics of a commercial grade item include those characteristics important to design, material, and performance of a commercial grade item that once verified, will provide reasonable assurance that the item will perform its intended safety function. The supplied parts with unacceptable dimensional variations is a deviation in the critical characteristics of the coupling's fabrication. In addition, the material choice and technical requirement to harden the material are questionable given the fact that corrosion was occurring in an environment with less chloride content than the original procurement specification. The material chosen with the heat treatment as specified are unacceptable from a design perspective. It has been concluded that the dimensional variations (fabrication deviation) and poor material selection (design deficiency) issues described herein require reporting under 10CFR Part 21.

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5) The date on which the information of such defect or failure to comply was obtained.

This condition was reported in Licensee Event Report, 2004-001, on July 21, 2004.

6) In the case of a basic component which contains a defect or fails to comply, the number and location of all such components in use at, supplied for, or being supplied for one or more facilities or activities subject to the regulations in this part.

Information received from Enertech lists the following companies/plants in the United States since 1975 supplied with this Goulds design coupling:

Company	Plant	Year Delivered
Mississippi Power and Light	Grand Gulf 1 & 2	1977
Consumers Power	Midland 1&2	1977-78
Cleveland Electirc Illuminating Co.	Perry 1 & 2	1978
Washington Public Power Supply System	WPPSS 1 & 4	1979
Washington Public Power Supply System	WPPSS 1 & 4	1979

7) The corrective action which has been, is being, or will be taken; the name of the individual or organization responsible for the action; and the length of time that has been or will be taken to complete the action.

See corrective actions section below.

8) Any advice related to the defect or failure to comply about the facility, activity, or basic component that has been, is being, or will be given to purchasers or licensees.

Review couplings of similar design and application in environments conducive to IGSCC. Verify adequate design margin for stress. Consider inspecting couplings that had installation difficulties as a result of out of specification tolerances.

CORRECTIVE ACTIONS

Corrective actions have been designed and implemented to increase the design margin, reduce potential installation impacts and ensure the appropriateness of the components delivered from suppliers. These corrective actions include:

Replacement of the ESW pump A (excluding the discharge head) with new components including the impellers, shafts and couplings. The couplings have been redesigned to provide a greater wall thickness and an improved material to resist corrosion. The new pump has a 4.5" diameter, 17-4 precipitation hardened (PH) stainless steel coupling with increased margin between the maximum and allowable shear stress at steady state and transient conditions. 17-4PH Stainless Steel is approximately five times more resistant to stress corrosion cracking than the previous 416 Stainless Steel.

The ESW B couplings were replaced with the improved design subsequent to the ESW A replacement.

The ESW C screwed couplings had been previously replaced with new keyed couplings prior to this event. The new coupling was manufactured from material with a lower hardness and therefore is less susceptible

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to IGSCC. Additionally, the keyway area of the ESW C pump coupling has more design margin than the ESW A pump coupling.

Requirements will be established to increase the level of FENOC oversight with commercial grade dedication by third party vendors.

A comprehensive risk-informed decision-making process will be developed and implemented.

A reliable method will be developed and implemented for the identification, tracking and management review of significant issues that includes, but does not rely on the corrective action program input.

Changes are being made to augment the management staff with top industry experienced personnel. Additionally, focused management decision-making training will be conducted.

PREVIOUS SIMILAR EVENTS

A review of PNPP LERs over the past 10 years identified one reportable condition that resulted from the failure of a pump coupling. LER 2003-004 documents the failure of the coupling on the same ESW A pump. At that time, it was recognized that the coupling material was susceptible to SCC and installation issues resulted in high stress and was considered the cause. The pump couplings were scheduled for replacement with the more resistant couplings at the time of this event.

A word search was performed to identify Condition Reports at the PNPP relating to pumps or coupling sleeves. This review did not identify any Martensitic or type 416 stainless steel couplings that failed due to stress corrosion cracking in raw water service. No evidence of a previous ESW pump coupling sleeve failure was found.

COMMITMENTS

No regulatory commitments were identified in this report.

Energy Industry Identification System Codes are identified in the text by square brackets [XX].